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Alternative Poloidal Field Configurations for ITER*

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Abstract—The US Home Team has investigated the physics and engineering issues for two alternate poloidal field coil configurations for ITER. The first is called the Segmented CS configuration, where all of the solenoid modules are pancake-wound. The second option, termed the Hybrid CS configuration, utilizes a layer-wound central module and pancake-wound end modules. Performance comparisons are presented for the baseline design and the two alternate PF configurations, characterizing the 21 MA reference scenario. Alternate operating modes such as reverse-shear operation and a 17 MA driven mode were evaluated, but are not reported here.

I. INTRODUCTION

The baseline configuration of the poloidal field (PF) coils for ITER features a 12 m long monolithic central solenoid, which is layer-wound and consists of three regions of varying superconductor grades. The stray field from the ends of this long coil inhibit plasma shaping flexibility and shape conformity of the plasma over the reference 21 MA ignited scenario. The plasma performance can be improved by vertically segmenting the solenoid. The Segmented CS configuration offers superior plasma performance but suffers from the necessity to invoke high field (~ 13 T) joints in the superconductor. Development of such high field joints is beyond the scope of the ITER Engineering Design Activity, and therefore this concept was abandoned. The difficulties of the Segmented CS design prompted the US Home Team to develop a Hybrid CS option, where the central portion of the solenoid is of the same design as the baseline design—only shorter. In this configuration, independently controlled end-coils are located beyond the central module (separated by vertical gaps of about 1.5 m) to allow space for the leads of the 9.5 m layer-wound central module. The Hybrid CS design suffers from excessive insulation bond stress but it is thought that a modest R&D effort can resolve this problem.

This paper addresses the performance of the baseline design and the two alternate concepts. Plasma control and power supply issues are addressed in [1], and the engineering aspects of the Hybrid CS design are presented in [2]. Details are available in the report of the US Home Team [3].

II. PF COIL CONFIGURATIONS

A. Detailed Design Configuration—the Baseline

Fig. 1 shows the PF configuration for the ITER Detailed Design Report [4], developed during 1996. The 12 m long CS coil is radially graded into three regions, a cost effective arrangement which also enhances poloidal flux production. The CS and two of the shaping coils (PF2 and PF7) require Nb₃Sn superconductor because of the high field levels. The other shaping coils are designed with NbTi superconductor. Coil specifications (current centroid, cross section, current capacity and conductor type) are listed in Table I.

Fig. 1. PF coil arrangement for the Detailed Design Report.

Table I. PF coil specifications for the Detailed Design Report.

coil name	R_c mm	Z_c mm	ΔR_c mm	ΔZ_c mm	NI_{cap} MA	cond. type
CS	2305	0	747	12073	140.9	Nb ₃ Sn
PF2	5859	9738	1554	1042	18.5	Nb ₃ Sn
PF3	12947	7277	766	963	9.4	NbTi
PF4	15364	-2090	593	2485	17.6	NbTi
PF5	13198	-8043	1046	2334	19.2	NbTi
PF6	9638	-9452	1180	1135	12.7	NbTi
PF7	5859	-9981	1580	1553	27.3	Nb ₃ Sn
PF8	15177	3809	273	915	3.5	NbTi

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B. Segmented CS Design

The Segmented CS design is shown in Fig. 2 and coil parameters in Table II. All of the solenoid modules (CS, CS1U&L, CS2U&L) are pancake-wound with Nb₃Sn superconductor, each with uniform current density. Coils PF2 and PF7, relieved by the end-coils, demand less current than in the Detailed Design configuration and are therefore designed with NbTi superconductor. The main CS module is about 9.5 m long, and gaps between the CS module and the end-coils are minimal. Leads must exit the solenoid modules radially and joints are located in the bore.

Fig. 2. PF coil arrangement for the Segmented CS Configuration.

Table II. PF coil specifications for the Segmented CS Configuration.

coil name	R_c mm	Z_c mm	ΔR_c mm	ΔZ_c mm	NI_{cap} MA	cond. type
CS0	2321	0	728	9455	108.9	Nb ₃ Sn
PF2	6049	9738	1175	1026	13.6	NbTi
PF3	12947	7277	766	963	9.4	NbTi
PF4	15364	-2090	593	2485	17.6	NbTi
PF5	13198	-7943	1115	1768	16.9	NbTi
PF6	9580	-9452	1295	1122	13.5	NbTi
PF7	5990	-9981	1057	1505	18.3	NbTi
PF8	15177	3809	273	915	3.5	NbTi
CS1U	2306	5672	759	1750	17.6	Nb ₃ Sn
CS1L	2306	-5672	759	1750	17.6	Nb ₃ Sn
CS2U	2300	7616	770	1998	21.4	Nb ₃ Sn
CS2L	2300	-7616	770	1998	21.4	Nb ₃ Sn

C. Hybrid CS Design

The Hybrid CS design features a 9.5 m central solenoid module, but it is layer-wound as in the baseline design and is therefore radially graded. Gaps of about 1.5 m high are provided between the CSM coil and the end-coils for the CSM leads. Although the layer-wound and radially graded coil produces more flux than the CS module of the segmented design, the gaps result in a loss of flux production. The end-coils (CSU and CSL) are pancake-wound.

Fig. 3. PF coil arrangement for the Hybrid CS Configuration.

Table III. PF coil specifications for the Hybrid CS Configuration.

coil name	R_c mm	Z_c mm	ΔR_c mm	ΔZ_c mm	NI_{cap} MA	cond. type
CSM	2335	0	760	9486	110.6	Nb ₃ Sn
PF1	3883	9774	645	1115	7.0	NbTi
PF2	5991	9774	1291	1115	14.2	NbTi
PF3	12974	7305	818	1018	9.9	NbTi
PF4	15214	386	292	1018	3.4	NbTi
PF5	15444	-2084	753	2099	16.9	NbTi
PF6	13194	-8163	986	2203	18.5	NbTi
PF7	9631	-9157	1194	1144	12.8	NbTi
PF8	5893	-9799	1513	1129	15.7	NbTi
PF9	3883	-9799	794	1129	7.7	NbTi
CSU	2294	7363	743	2098	19.3	Nb ₃ Sn
CSL	2294	-7410	743	2194	19.3	Nb ₃ Sn

III. THE ISSUES

Vertically segmenting the ITER central solenoid offers improved plasma shape conformity over the scenario and enables higher upper triangularity, but results in poloidal flux crossing the straight portion of the TF coil inner leg which increases out-of-plane loads in this region. Segmentation also introduces gaps or inhibits radial grading in the CS which reduces poloidal flux consumption. Reducing the vertical extent of the CS coil, however, requires less compensation currents from the PF coils. This enables all of the PF coils to be designed with NbTi superconductor which is a significant cost savings over Nb₃Sn conductor.

IV. REFERENCE SCENARIO PERFORMANCE

The assessment of performance aspects were conducted with the Corsica [5] tokamak modeling code, particularly its free-boundary equilibrium solver and coil diagnostics modules.

A. Requirements

The 21 MA ignited mode of operation is the design basis for the ITER PF system. This operating mode is characterized by

a set of nominal plasma parameters, and associated ranges of parameters which indirectly determine the cross-sectional size of the poloidal field coils. With the coils established for this operating mode, the system is subsequently characterized for other operating modes.

Requirements on the poloidal field system include shape tolerances which are applicable during the power-producing portion of the scenario, from the start of the current flattop state through the end-of-burn state. These tolerances are:

1. maintain the separatrix (and strike-points) in the divertor channels to within ± 100 mm of their reference position,
2. maintain the outer edge of the plasma in the region facing an ion-cyclotron heating (ICH) antenna to within ± 50 mm of its nominal position (without exceeding a maximum radius of 10960 mm), and
3. maintain a vacuum gap between the first-wall and the plasma scrape-off-layer (SOL) envelope of no less than 100 mm.

The SOL envelope is defined as the flux surface passing through a point 50 mm outboard of the plasma edge. The tolerances listed above are static requirements, excursions beyond these values are allowable during transient control of disturbances.

The PF system is required to produce 1000s burn duration (about 80 Wb of burn flux capability) for the 21 MA reference scenario, within the current and field limits of the CS.

B. Operational Flexibility

The important plasma parameters are beta and internal inductance. The poloidal beta corresponding to 1500 MW plasma power is:

$$\beta_p \equiv \frac{4V\langle p \rangle}{\mu_0 R_0 I_p^2} = 0.9, \quad (1)$$

where V is the plasma volume, p the kinetic pressure, R_0 is the geometric mean radius (major radius), I_p is the toroidal current and μ_0 the permeability of free-space. The symbol pair $\langle \rangle$ denote a volume average over the plasma. The normalized internal inductance is defined as:

$$\ell_i \equiv \frac{\langle B_p^2 \rangle}{\mu_0^2 R_0 I_p^2} = 0.9, \quad (2)$$

where B_p is the poloidal magnetic field. Operational flexibility is expressed with ranges in beta and internal inductance: $0.7 \leq \beta_p \leq 1.2$, and $0.7 \leq \ell_i \leq 1.1$, where the beta range is applicable over the burn phase ($\beta_p = 0.1$ at the SOF state).

C. Flux States

Flux states are established either by coil technology limits, e.g., the end-of-burn (EOB) state occurs when the CS reaches

a peak field of 12.75 T, or by inductive and resistive flux consumption during the ramp-up, which set the start-of-flattop (SOF) and start-of-burn (SOB) states. Resistive flux consumption is estimated with Ejima scaling [6]:

$$\Delta \Psi_{resistive} \approx C_E \mu_0 R_0 I_p \quad (3)$$

with a coefficient of $C_E = 0.45$ which establishes the flux linkage at the start-of-flattop state. Resistive consumption during the heating phase (10 Wb) is based on a loop voltage of 0.2 V over a 50 s heating period.

D. Shape Conformity Results

Fig. 4 shows the plasma boundaries over the full $\beta_p - \ell_i$ range of the 21 MA scenario at the three fiducial flux states. Shape deviations are summarized in Table IV.

Fig. 4. Plasma boundary overlays at the SOF, SOB and EOB flux-states over the full range of operational space for the 21 MA scenario from the baseline (DDR) and the two alternate coil sets (SCS and HCS).

Table IV. Average maximum separatrix deviations (mm) in the inboard divertor channel (#1), outboard divertor channel (#2) and ICH antenna region (#3) over the full range of operational space for the 21 MA scenario from the baseline (DDR) and the two alternate coil sets (SCS and HCS).

region	DDR			SCS			HCS		
	#1	#2	#3	#1	#2	#3	#1	#2	#3
SOF	38	73	34	28	58	40	18	28	16
SOB	44	36	13	15	21	11	7	9	8
EOB	50	28	11	45	34	15	19	26	11

E. Plasma Parameters

The burn flux capability: $\Delta \Psi_{burn}$, MHD safety factor: q_{95} , upper triangularity: δ_{U95} , and average triangularity: δ_{95} , during the current flattop portion of the nominal 21 MA scenario are presented in Table V.

Table V. Plasma parameters for the reference 21 MA scenario with nominal plasma parameters for the three coil set options.

		DDR	SCS	HCS
$\Delta \Psi_{burn}$	(Wb)	80.6	77.8	79.4
q_{95}	SOF	2.83	3.00	2.94
q_{95}	SOB	2.94	3.07	3.07
q_{95}	EOB	3.04	3.04	3.08
δ_{U95}	burn	0.25	0.28	0.30
δ_{95}	burn	0.23	0.24	0.25

F. Coil Current Demands

Peak coil current demands over the operational space of the 21 MA scenario are compared in Table VI for the three coils sets.

Table VI. Coil current demands over the full operational space of the 21 MA scenario from the three coil sets. (PF2 from initial magnetization state.)

	DDR	SCS	HCS	
CS	138.6	109.1	111.1	CSM
PF2	16.0	12.1	7.0	PF1
PF3	7.5	7.8	14.2	PF2
PF8	—	—	9.4	PF3
PF4	17.6	17.2	—	PF4
PF5	17.6	15.8	14.4	PF5
PF6	9.9	13.5	18.0	PF6
PF7	27.3	13.2	10.1	PF7
CS1U	—	15.3	15.7	PF8
CS1L	—	16.8	7.7	PF9
CS2U	—	21.4	19.3	CSU
CS2L	—	21.1	19.3	CSL

G. Out-of-Plane Loads

Fig. 5 shows a comparison of the out-of-plane loads on the TF coil winding pack at the start-of-flattop state for the baseline and two alternate coil sets.

Fig. 5. Out-of-plane loads on the TF coil winding packs at the SOF state.

It is interesting that the *peak* loading, occurring near the lower divertor coils ($24 \leq s \leq 30$ m), is greatest for the Detailed Design configuration, but in the straight-leg region the loading is greater for both the Segmented and Hybrid CS options. The out-of-plane loads in the straight section of the TF coil get transmitted to the CS or CSM coils producing a torsional stress in the solenoid winding. Some of this loading is taken up by the insulation bond around the conductors, and in the case of the Hybrid CS in particular, the insulation bond stress exceeds allowable values by almost a factor-of-two.

H. 21 MA Scenario Performance Summary

Fig. 4 and Table IV clearly show the benefit of a shorter central solenoid. Although all three coil sets satisfy the design

requirements, the Segmented and Hybrid CS designs offer significant reduction in shape deviations in the divertor channels and antenna region. The alternate coil sets also improve the MHD safety factor and triangularity (Table V). There is some loss in poloidal flux production in the alternate designs, but with an estimated loop voltage of 0.07 V or less during the burn phase, all designs will provide more than 1000 s of burn duration.

V. FINAL DESIGN CONFIGURATION

The high-field joint technology demanded by the Segmented CS design, and the excessive insulation bond stress in the CSM module in the Hybrid CS design resulted in abandonment of these two alternate coil configurations. The configuration chosen for the ITER Final Design [7] is essentially that of the Detailed Design Report, except the primary elongation coils (PF2 and PF7 in Fig. 1) have been split into two coils to enable their construction with NbTi superconductor. This offers some cost savings, although little improvement in performance.

VI. SUMMARY

Both the Hybrid CS and Segmented CS designs offer significant reductions in plasma shape deviations, and both offer some cost savings. The Segmented CS design requires 13 T superconducting joint technology which is beyond the scope of the EDA, but the Hybrid concept may be feasible if insulation R&D is successful.

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